NON-PUBLIC?: N
ACCESSION #: 9106240321
LICENSEE EVENT REPORT (LER)

FACILITY NAME: DIABLO CANYON UNIT 1 PAGE: 1 OF 7

DOCKET NUMBER: 05000275

TITLE: REACTOR TRIP DUE TO PERSONNEL ERROR AND SAFETY INJECTION DUE TO
LEAKING STEAM DUMP VALVES
EVENT DATE: 05/17/91 LER #: 91-009-00 REPORT DATE: 06/14/91

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv) and
50.73(a)(2)(i)(B)

LICENSEE CONTACT FOR THIS LER:
NAME: MARTIN T. HUG - SENIOR REGULATORY TELEPHONE: (805) 545-4005
COMPLIANCE ENGINEER

COMPONENT FAILURE DESCRIPTION:
CAUSE: B SYSTEM: SB COMPONENT: V MANUFACTURER: C635
REPORTABLE NPRDS: N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On May 17, 1991, at 0624 PDT, with Unit 1 operating at 100 percent power, a reactor trip occurred due to nuclear instrumentation power range two-out-of-four channels high flux at high setpoint signals: At 0625 PDT, a safety injection occurred due to two-out-of-four low pressurizer pressure signals. During the event, reactor coolant system cooldown exceeded the allowable rate of 100 degrees Fahrenheit per hour of Technical Specification 3.4.9.1.b. An Unusual Event was declared at 0625 PDT on May 17, 1991. A one-hour emergency report required by 10 CFR 50.72(a)(1)(i) was made on May 17, 1991, at 0633 PDT.

The cause of the reactor trip was determined to be personnel error. An I&C technician inadvertently deenergized a second nuclear instrumentation power range channel (N42) while performing surveillance testing on another power range channel (N41). The cause of the safety injection was steam dump valves that failed open and overcooled the reactor coolant system.

Corrective actions for the event included: (1) temporary stoppage of all I&C work until I&C personnel were tailboarded on the necessity of self-verification; and (2) installation of switch and fuse covers on each channel to act as a physical barrier to prevent inadvertent actions.

Corrective actions for the steam dump valves will be discussed in Licensee Event Report 1-90-017-01.

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END OF ABSTRACT



I. Plant Conditions

Unit 1 was in Mode 1 (Power Operation) at 100 percent power.

II. Description of Event

A. Event:

On May 17, 1991, at 0620 PDT, two Instrumentation and Controls (I&C) technicians were performing Surveillance Test Procedure (STP) I-2D, "Nuclear Power Range Incore/Excore Calibration," on nuclear instrumentation (NI) (IG) power range channel N41. The instrument channel under test had the cabinet instrument drawer (IG) (PL) open. The next step in the procedure required removal of the fuse (IG) (FU) for reconnection of that channel's signal and high voltage cables. The technician who was to remove the fuse had previously manipulated the fuse in that channel while the drawer was closed. The technician went to the instrument drawers and inadvertently pulled the fuse for NI channel N42, which had its drawer closed and was adjacent to channel N41. This resulted in the protection logic recognizing a two-out-of-four high flux at high power signal coincidence for a reactor (AB)(RCT) trip.

On May 17, 1991, at 0624 PDT, with Unit 1 at 100 percent power, a reactor trip occurred due to the protection system two-out-of-four high flux at high setpoint signal coincidence, which initiated a main turbine (TA)(TRB) trip. Following the reactor and main turbine trips, the condenser steam dump valves (SDV)(SB)(V) automatically opened to prevent reactor coolant system (RCS) (AB) pressure and temperature increase. RCS pressure and temperature decreased and, to close the SDVs and mitigate the cooldown, a close signal was manually initiated. Two SDVs, 1-PCV-1 and 1-PCV-11, did not close following the close signal.

Control room operators entered Emergency Procedure E-0, "Reactor Trip or Safety Injection," and verified the automatic responses of the protection system.

One minute and 38 seconds after the reactor trip, RCS pressure and temperature had decreased sufficiently to result in a safety injection (SI) on two-out-of-four low pressurizer (AB) (PZR) pressure (

An Unusual Event (UE) was declared on May 17, 1991, at 0625 PDT, in response to the SI. Due to previous experience with SDVs failing to close following actuation, operators quickly identified the malfunction. At 0627 PDT, on May 17, 1991, operators terminated the





cooldown by closing the main steam isolation valves (MSIV)(SB)(V) and MSIV bypass valves (SB) (V).

A one-hour emergency report required by 10 CFR 50.72(a)(1)(i) was made on May 17, 1991, at 0633 PDT. This report indicated that RCS pressure dropped from the normal operating pressure (NOP) of 2235 psig to 1640 psig and T sub avg dropped from the normal operating temperature (NOT) of 574 degrees to 507 degrees. Subsequent analysis of recorded data determined that RCS pressure and temperature had reached as low as 1730 psig and 465 degrees T sub avg, exceeding Technical Specification 3.4.9.1.b. limits for RCS cooling of 100 degrees in one hour.

On May 17, 1991, at 0800 PDT, operators returned Unit 1 to NOP and NOT, and stabilized the Unit in Mode 3 (Hot Standby).

B. Inoperable Structures, Components, or Systems that Contributed to the Event:

None.

C. Dates and Approximate Times for Major Occurrences:

1. May 17, 1991, at 0624 PDT: Event/discovery date. Unit 1 reactor trips when a second power range NI channel is inadvertently deenergized by an I&C technician, which satisfies the two-out-of-four protection system logic for a reactor trip.

2. May 17, 1991, at 0625 PDT: An SI results due to RCS pressure decreasing caused by two SDVs leaking excessively following automatic open actuation and manual close signal. A UE is declared.

3. May 17, 1991, at 0627 PDT: Operators manually close the MSIV and MSIV bypass valves to isolate the SDVs.

4. May 17, 1991, at 0633 PDT: A one-hour emergency report required by 10 CFR 50.72(a)(1)(i) was made.

5. May 17, 1991, at 0800 PDT: Unit 1 was stabilized in Mode 3 at NOP and NOT.





D. Other Systems or Secondary Functions Affected:

Other than the two SDVs, all equipment functioned as intended to stabilize Unit 1 in Mode 3.

Valves 1-PCV-1 and 1-PCV-11 opened per design but, following closure signal, failed to close. It was determined that t
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inner plug stem of the valves separated from the main stem, and resulted in excessive leakage through the valves. The valve stem separation was the result of microwelding of the valve seat. Corrective actions being taken regarding the SDVs failure to close will be discussed in LER 1-90-017-01.

E. Method of Discovery:

The reactor trip was immediately apparent to plant operators due to alarms and indications received in the control room. The cause of the SI (SDV failure) was apparent to the operators due to a recent similar transient.

F. Operators Actions:

Operators verified the automatic functions of the protection system. Operators closed the SDVs, but the cooldown continued. Operators then closed the MSIV and MSIV bypass valves terminating the cooldown of the RCS.

G. Safety System Responses:

1. The reactor trip breakers (AA)(BKR) opened.

2. The control rod drive mechanism (AA)(DRIV) allowed the control rods to drop into the reactor.

3. The main turbine tripped.

4. An SI signal was initiated on low pressurizer pressure.

5. The SI pumps (BQ)(P) started.

6. The residual heat removal pumps (BP) (P) started.

7. The charging pumps (BQ)(P) started.

8. The motor-driven auxiliary feedwater (AFW) pumps (BA)(MO)(P) started per design.

9. Diesel Generators (EK)(DG) 1-1, 1-2, and 1-3 started, and per design did not load.

III. Cause of the Event

A. Immediate Cause:

An I&C technician, while performing surveillance testing on NI power range channel N41, inadvertently removed a fuse in NI power range channel N42.

B. Root Cause:

The root cause was determined to be personnel error (cognitive) in that the I&C technician did not perform self-verification. I&C policy, "Policy For Unit/Channel/Component Self-verification," dated June 30, 1988, requires that an individual verify his own action as correct prior to performing the action.

IV. Analysis of the Event

A. Safety Analysis:

1. Reactor Trip and Safety Injection:

Impact to the departure from nucleate boiling ratio (DNBR) resulting from accidental depressurization of the RCS caused by secondary side depressurization is analyzed in Section 15.2.13 of the Final Safety Analysis Report (FSAR) Update. Accidental depressurization of the RCS is identified as a Condition II event (Faults of Moderate Frequency). The overcooling and resulting depressurization of the RCS reported in this LER is bounded by the FSAR Update analysis, and therefore the minimum DNBR was not lower than 1.30 during this event.

2. Overcooling:

A Westinghouse engineering evaluation of the RCS considered the impact of the thermal transient upon the pressurizer, reactor vessel, RCS piping, the thick metal of the steam generators (AB)(SG), and the reactor coolant pumps (AB)(P). Westinghouse reviewed temperature and pressure data and compared these with evaluations of similar transients at other plants and the evaluation performed for PG&E's December 24, 1990 rapid cooldown event (LER 1-90-017-00). The comparison showed that the

DCPP Unit 1 rapid cooldown event is bounded by transients previously analyzed. The Westinghouse evaluation concluded that the above described DCPP Unit 1 transient did not adversely affect the structural integrity of the affected components and system, and that the RCS could be returned to NOP and NOT and the unit restarted safely.

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V. Corrective Actions

A. Immediate Corrective Actions:

1. The plant was stabilized in Mode 3.

2. An immediate, temporary I&C work stoppage was directed by management. I&C personnel were tailboarded on the necessity of self-verification.

B. Corrective Actions to Prevent Recurrence:

1. A temporary physical barrier has been installed over the NI drawer face to prevent inadvertent operation of switches or fuse activity.

2. A memorandum has been sent to Shift Control Technicians informing them of the new temporary physical barriers and their use.

3. A design change will be implemented to install a permanent, removable physical barrier design for the NI drawers.

4. In addition to the self-verification training already part of the Maintenance Department training, additional self-verification training will be integrated into I&C laboratory performance training.

5. The technician involved in the event has been counselled as to the necessity for self-verification.

6. An INPO video on self-verification practices will be presented to I&C personnel during the quarterly update meeting.

## VI. Additional Information

### A. Failed Components:

Copes Vulcan valves 1-PCV-1 and 1-PCV-11, model D-100-160-3, eight inch.

### B. Previous Similar LERs:

1. ESF Actuations Due To Personnel Error:

LER 1-91-005-00, "Actuation of Wrong Test Switch Causes Unplanned Diesel Generator Start (ESF) Actuation due to Personnel Error," describes an event wherein a non-licensed operator inadvertently actuated the wrong Solid State Protection System (SSPS)(JG) test





switch resulting in an unplanned Emergency Diesel Generator start, an ESF actuation. The root cause of this event was failure to follow self-verification policies. The corrective action did not prevent the most recent event reported in LER 1-91-009-00 since the counselling was directed only at Operations personnel.

2. Condenser 40 Percent Steam Dump Valves:

LER 1-90-017-00, dated January 23, 1991, reported a Unit 1 reactor trip and SI due to a stuck open pressurizer spray valve. During this event, SDV 1-PCV-1 leaked excessively following actuation. The corrective actions to prevent recurrence for this event dealt primarily with the stuck open pressurizer spray valve, which would not have prevented the contributory cause on the SDVs failure reported in LER 1-91-009-00. The corrective actions to prevent SDV failure have identified but had not been completed on Unit 1 at the time of the May 17, 1991 reactor trip. These corrective actions will be described in LER 1-90-017-01.

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James D. Shiffer
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Nuclear Power Generation

June 17, 1991

PG&E Letter No. DCL-91-154

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80

Diablo Canyon Unit 1
Licensee Event Report 1-91-009-00
Reactor Trip Due to Personnel Error and Safety Injection Due to
Leaking Steam Dump Valves

Gentlemen:

Pursuant to 10 CFR 50.73(a)(2)(iv) and 10 CFR 50.73(a)(2)(i)(B), PG&E is submitting the enclosed Licensee Event Report (LER) concerning a reactor trip resulting from personnel inadvertently satisfying reactor trip logic while performing surveillance testing. During the recovery from the reactor trip, the reactor coolant system was cooled at a rate greater than 100 degrees fahrenheit in any hour, in violation of Technical Specifications.

This event has in no way affected the health and safety of the public.

Sincerely,

J. D. Shiffer

cc: Ann P. Hodgdon
John B. Martin
Phillip J. Morrill

Paul P. Narbut
Harry Rood
CPUC
Diablo Distribution
INPO

DC1-91-TI-N047

Enclosure

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